10CFR50.73

Virginia Electric and Power Company
North Anna Power Station
P. O. Box 402
Mineral, Virginia 23117

February 15, 2002

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Serial No.: 02-035
NAPS: JHL
Docket No.: 50-339
License No.: NPF-7

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Power Station Unit 2.

Report No. 50-339/2001-005-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,

DAH

D. A. Heacock, Site Vice President
North Anna Power Station

Enclosure

Commitments contained in this letter: None

cc: United States Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23 T85
Atlanta, Georgia 30303-8931

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

IE22

Rec'd 03/05/02

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
NORTH ANNA POWER STATION , UNIT 2	05000 - 339	1 OF 5

TITLE (4)

Automatic Reactor Trip Due To Turbine Control System Power Supply Failure

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCUMENT NUMBER 05000-
12	22	2001	2001	-- 005 --	00	02	15	2002	FACILITY NAME	DOCUMENT NUMBER 05000-

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)							
			20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
POWER LEVEL (10)	100 %		20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
D. A. Heacock, Site Vice President	(540) 894-2101

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	EC	RECT		N						
X	EC	IS	L045	N						

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) **(16)**

On December 22, 2001, at 1542 hours, with Unit 2 in Mode 1 operating at 100% power, an automatic reactor trip occurred due a to failure in the turbine control electro hydraulic control (EHC) power supply system. A momentary fault on the control system power supply created a situation where the EHC system controller changed operating modes and reset its demand to zero. Consequently, the turbine valves drifted closed. The loss of load transient caused a reactor trip on low-low steam generator level. The reactor trip signal initiated a turbine trip. A non-emergency 4-hour notification was made to the NRC, at 1920 hours, on December 22, 2001, in accordance with 10CFR50.72(b)(2)(iv)(B). During the event, the auxiliary feedwater (AFW) system and accident mitigation system actuation circuitry (AMSAC) actuated. A non-emergency 8-hour notification was also made to the NRC, at 1920 hours, on December 22, 2001, in accordance with 10CFR50.72(b)(3)(iv)(A). The cause of the event was a failure of components in both the normal and backup turbine control EHC power supplies. Repairs to the components in both the normal and backup turbine control EHC power supply were performed prior to unit startup. No significant safety consequences resulted from this event because the reactor protection system and ESF systems functioned as designed following the reactor trip. The health and safety of the public were not affected at any time during this event.

1.0 DESCRIPTION OF THE EVENT

On December 22, 2001, at 1542 hours, with Unit 2 in Mode 1 operating at 100% power, an automatic reactor trip occurred due to a failure in the turbine control electro hydraulic control (EHC) power supply system. A momentary fault on the control system power supply created a situation where the EHC system controller changed operating modes and reset its demand to zero. Consequently, the turbine valves drifted closed. The loss of load transient caused a reactor trip on low-low steam generator level. The reactor trip signal initiated a turbine trip. A more detailed description of the event follows.

On December 22, 2001, North Anna Unit 2 had been operating at 100% power and was on-line for 8 days following a maintenance outage. No significant equipment was out of service. On December 22, 2001, at 1541 hours, the main control room received a main control board "first out" annunciator for a "EHC DC Power Supply Failure-Turbine Trip". A momentary fault on the EHC control system (EIIS System TG) power supply (EIIS Component JX) caused turbine trip block protection solenoid operated valve (SOV) (EIIS Component PSV), 20-AST-1, to briefly energize. However, the main turbine (EIIS System TA) did not trip because the momentary fault was not of sufficient duration to allow auto stop oil pressure to decrease below the 45 psig setpoint required to generate a turbine trip signal to the reactor protection system (EIIS System JC). The disturbance on the power supply created a situation where the EHC system controller changed operating modes and reset its demand to zero. Following the receipt of the initial turbine trip annunciator, the turbine valves started to close; resulting in a loss of load. This resulted in increasing steam header pressure with a subsequent shrink in steam generator levels. Approximately 5 seconds later, a reactor trip was initiated automatically from a valid low-low level in the "A" Steam Generator (EIIS System AB, Component SG). The reactor trip signal initiated a main turbine trip.

Control Room personnel responded to the reactor trip in accordance with emergency procedure 2-E-0, Reactor Trip or Safety Injection. The post trip response progressed as expected and the Operators transitioned to 2-ES-0.1, Reactor Trip Response. All Engineered Safety Feature (ESF) equipment responded as designed.

Coincident with the reactor trip, Reactor Coolant System (RCS) pressure increased to approximately 2353 psig due to the loss of load caused by the turbine valves going closed. The pressurizer power operated relief valves (PORVs) (EIIS Component RV) opened to reduce RCS pressure. When RCS pressure decreased below the PORV lift setpoint, the pressurizer PORVs closed as designed. The pressure increase is expected following a loss of load from 100% power. RCS pressure then decreased to approximately 1940 psig and RCS temperature decreased to approximately 546 °F, before recovering to the "no-load" values of 547 °F and a pressure of 2235 psig. The steam dumps functioned normally in T_{avg} mode. The pressure drop is expected during a reactor trip due to the temperature-dependent shrink of the primary system following the

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
NORTH ANNA POWER STATION	05000 - 339	2001	--005 --	00	3 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

reactor trip. Pressurizer level dropped to approximately 23.5 percent before recovering to its no-load value of 28 percent. The level drop is normal and expected - a consequence of the temperature-induced shrinkage following any reactor trip. Recovery of pressurizer level was within the capability of the normal letdown and charging alignment.

A non-emergency four-hour report was made to the NRC Operations Center, at 1920 hours, on December 22, 2001, pursuant to 10CFR50.72(b)(2)(iv)(B) for any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical. During the event, the auxiliary feedwater (AFW) system (EIIS System BA) and accident mitigation system actuation circuitry (AMSAC) actuated as designed. A non-emergency 8-hour notification was also made to the NRC, at 1920 hours, on December 22, 2001, in accordance with 10CFR50.72(b)(3)(iv)(A) for any event or condition that results in valid Engineered Safety Function actuation.

Unit equipment responded as expected with a few discrepancies. The discrepancies included: 1) the "A" Reactor Coolant Pump, 2-RC-P-1A, (EIIS Component P) vibration alarm was received during the transient but cleared when acknowledged, 2) three secondary side relief valves lifted and failed to reseat, 3) limit indication for condensate recirculation flow control valve, 2-CN-FCV-207, was not indicating properly, and 4) the individual rod position indicator (IRPI) (EIIS System AA, Component ZI) for control rod K12 or K14 did not indicate correctly.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event posed no significant safety implications because the reactor protection system and ESF systems functioned as designed following the reactor trip. Therefore, the health and safety of the public were not affected by this event.

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) for any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF) including the reactor protection system (RPS).

3.0 CAUSE

North Anna Units 1 and 2 use a EHC control system that can be powered from the "primary" source, a 120VAC vital bus, or the "secondary" source, the main turbine permanent magnet generator (PMG) output (only available when the turbine is spinning at sufficient speed). Assuming both are available, a loss of either source should not cause a loss of control power due to an automatic transfer to the redundant power supply. The cause of the automatic reactor trip was a failure of components in both the normal and backup turbine control EHC power supplies. Troubleshooting activities following the reactor trip identified the failure of auctioneering diode (EIIS Component RECT) D4 on the +15 volt DC (EIIS System EC) primary power supply circuit. The diode appeared to be open preventing the output of the power supply from reaching the 15-volt DC power bus.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

It was also identified that test switch (EIIS Component IS) SW5-B in the secondary power supply had an intermittent open preventing the secondary power supply current to pass to the bus. The switch was found to have a loose rivet.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Control Room personnel responded to the reactor trip in accordance with emergency procedure 2-E-0, Reactor Trip or Safety Injection. The lowest RCS pressure during the event was 1940 psig and the lowest RCS temperature was 546 degrees F. Pressurizer level dropped to approximately 23.5 percent. Pressurizer pressure, pressurizer level, and RCS temperature returned to normal programmed values. All ESF equipment responded as designed.

The post trip response progressed as expected and the Operators transitioned to 2-ES-0.1, Reactor Trip Response. The plant was stabilized at no-load conditions.

5.0 ADDITIONAL CORRECTIVE ACTIONS

A Post Trip Review meeting was conducted, on December 22, 2001, to identify the cause of the reactor trip to prevent recurrence, to identify abnormal or degraded indications occurring during the reactor trip, and to assess Unit readiness for return to operation.

The "A" Reactor Coolant Pump, 2-RC-P-1A, vibration alarm was received during the transient but cleared when acknowledged. Subsequent investigation showed normal vibration levels for all three reactor coolant pumps.

Repairs were performed on the secondary side relief valves to get them to reseat.

Limit indication for 2-CN-FCV-207 was verified as correct by local valve position. The problem was determined to be a burned out light bulb, which was subsequently replaced.

One of the Individual Rod Position Indicators (IRPIs), K12 or K14, located in Control Bank "B" and Control Bank "A" respectively did not initially decrease to 0 steps (the operator at the controls could not remember which IRPI stuck). Immediately following the trip, the reactor operator tapped the stuck IRPI causing it to deflect to the correct indication of 0 steps. Maintenance verified satisfactory operation of K-12 and replaced K-14's indicator. Both IRPIs were then calibrated satisfactorily and returned to service.

6.0 ACTIONS TO PREVENT RECURRENCE

A root cause evaluation is being performed regarding the automatic reactor trip. Corrective actions will be performed as necessary following completion of the evaluation.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
NORTH ANNA POWER STATION	05000 - 339	2001	--005 --	00	5 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

7.0 SIMILAR EVENTS

LER 50-338/86-002-00 documents a reactor trip, from 100%, power generated by a low-low level in "B" steam generator caused by closure of the turbine governor valves. The closure of the turbine governor valves was attributed to problems associated with the control system.

LER 50-338/89-014-00 documents a reactor trip from, 90% power, due to loss of electro hydraulic control (EHC) system pressure. The turbine trip solenoid operated valve 20-ET o-ring failed.

LER 50-338/89-017-00 documents a reactor trip, from 7% power, generated by a low-low level in "B" steam generator caused by electro hydraulic control (EHC) system pressure transients. The EHC system pressure transient was caused by leaking turbine overspeed protection circuitry (OPC) valves.

8.0 MANUFACTURER/MODEL NUMBER

The 15 amp auctioneering diode that failed was manufactured by Powerex Inc. (PRX), Part No. 368C.

The test switch that failed was a heavy duty double-pole, single throw switch manufactured by Eaton Commercial Controls Division, Catalog No. 7310K38.

9.0 ADDITIONAL INFORMATION

North Anna Unit 1 was in Mode 1 at 100% power and was not affected by this event.